UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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April 13, 2001

Harold B. Ray, Executive Vice President
Southern California Edison Co.
San Onofre Nuclear Generating Station
P.O. Box 128
San Clemente, California 92674-0128

SUBJECT: NRC INTEGRATED INSPECTION REPORT 50-361/01-03; 50-362/01-03

Dear Mr. Ray:

On March 31, 2001, the NRC completed an inspection at your San Onofre Nuclear Generating Station, Units 2 and 3. The enclosed report documents the inspection findings which were discussed on March 2, 8, and 23 and April 3, 2001, with Mr. R. Krieger and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Circumstances affecting the financial viability of Southern California Edison Co. have continued to evolve during this inspection period. Actions have been initiated by the state of California and Southern California Edison Co. to address the impacts of these financial challenges. The NRC has exercised communications channels to better understand your planned and implemented actions, especially as they relate to your responsibility to safely operate the San Onofre reactors. NRC inspections, to date, have confirmed that you are operating these reactors safely and that public health and safety are, thus far, assured.

In response to these conditions of economic stress, there are two differences in how the Region communicates its inspection findings. First, we will continue the 6-week periodicity of our integrated inspection reports (the other reactors in Region IV transitioned to a quarterly report frequency, with the exception of Diablo Canyon). Second, the description of the scope of the individual inspection activities will be significantly more detailed. This is being done to keep the public more fully informed of the breadth and depth of the NRC's inspection and oversight activities.

Based on the results of this inspection, the NRC has identified two issues that were evaluated under the risk Significance Determination Process as having very low safety significance (Green). The NRC has also determined that a violation is associated with one of these issues. This violation is being treated as a noncited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy. The NCV is described in the subject inspection report. If you contest the violation or significance of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the San Onofre Nuclear Generating Station, Units 2 and 3, facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if any, will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

***/RA/***

Charles S. Marschall, Chief
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Dockets: 50-361
50-362
Licenses: NPF-10
NPF-15

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NRC Inspection Report
50-361/01-03; 50-362/01-03

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NRR Event Tracking System **(IPAS)**
SONGS Site Secretary **(SFN1)**
Dale Thatcher **(DFT)**

| RIV:RI:DRP/C | SRI:DRP/C | C:DRS/PSB | C:DRS/EMB | C:DRP/C |
|---|---|---|---|---|
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| 4/12/01 | Not Available | 4/10/01 | 4/11/01 | 4/ /01 |

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**ENCLOSURE**

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

| | |
|---|---|
| Dockets: | 50-361<br>50-362 |
| Licenses: | NPF-10<br>NPF-15 |
| Report No.: | 50-361/01-03<br>50-362/01-03 |
| Licensee: | Southern California Edison Co. |
| Facility: | San Onofre Nuclear Generating Station, Units 2 and 3 |
| Location: | 5000 S. Pacific Coast Hwy.<br>San Clemente, California |
| Dates: | February 18 through March 31, 2001 |
| Inspectors: | J. A. Sloan, Senior Resident Inspector<br>J. G. Kramer, Resident Inspector<br>J. B. Nicholas, Senior Health Physicist<br>C. J. Paulk, Senior Reactor Inspector |
| Accompanying Personnel: | J. L. Taylor, Reactor Inspector |
| Approved By: | C. S. Marschall, Chief, Project Branch C |

SUMMARY OF FINDINGS

San Onofre Nuclear Generating Station, Units 2 and 3
NRC Inspection Report 50-361/01-03; 50-362/01-03

IR05000361-01-03, IR05000362-01-03: 02/18-03/31/2001; Southern California Edison; San Onofre Nuclear Generating Station, Units 2 & 3; Integrated Resident and Regional Report; Heat Sink Performance, Event Followup, Other.

Resident and region-based reactor and radiation inspectors conducted the inspection. This inspection identified two Green findings, one of which was a noncited violation with two examples. The significance of all issues is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, “Significance Determination Process.”

Cornerstone: Mitigating Systems

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified, with two examples, for having inadequate measures to assure that the design basis of the shutdown cooling heat exchangers, and safety-related room coolers supplied by the emergency chilled water system, were correctly translated into procedures or were maintained, respectively. This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This violation was entered into the licensee’s corrective action program as Action Requests 000401144 and 010300419.

  The violation was of very low safety significance because: (1) there were no audible indications of damage to the shutdown cooling heat exchangers and there was not a history of leaks for the heat exchangers; and (2) the total emergency chilled water flow exceeded the system design basis and the preliminary test data, along with calculations, provide assurance that adequate flow can be supplied to each safety-related room cooler (Sections 1R07 and 4OA5.3).

- Green. The licensee reported (Licensee Event Report 361; 362/2000-010-00) that a cracked weld on the electrical power supply conduit coupling connection for the Train B control room emergency air cleanup system recirculation fan could prevent the train from performing its safety function during a seismic event. The issue is in the licensee’s corrective action program as Action Request 000801751.

  The issue was of very low safety significance, because only one cornerstone was involved, only one train was affected, and there was no actual loss of safety function (Section 4OA3).

# Report Details

Summary of Plant Status:

Unit 2 operated at essentially 100 power throughout this inspection period.

Unit 3 operated in Mode 5 throughout this inspection period, conducting repairs to nonsafety-related electrical and turbine systems damaged during a February 3, 2001, event (NRC Inspection Reports 50-361; 362/01-02 and 50-362/01-05).

## 1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

### 1R04 Equipment Alignments (71111.04)

a. Inspection Scope

The inspectors performed partial walkdowns during outages of the following systems to confirm the operability of the redundant trains:

- Auxiliary feedwater pump 2P504 (Unit 2)
- Charging pump 2P192 (Unit 2)
- Emergency Diesel Generator (EDG) 3G002 (Unit 3)

The inspectors checked component oil levels, valve and electrical lineups, and control room indications. When appropriate, the inspectors observed proper operation of redundant trains of equipment.

b. Findings

No findings of significance were identified.

### 1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors performed routine fire inspection tours for the following plant areas important to reactor safety:

- Train A emergency core cooling system pump room (Unit 2)
- Train B emergency core cooling system pump room (Unit 2)
- Swing high pressure safety injection Pump 2P018 pump room (Unit 2)
- Train A component cooling water surge tank room (Unit 2)
- Train B component cooling water surge tank room (Unit 2)
- Train A Class 1E 4160 volt switchgear room (Unit 2)
- Train B Class 1E 4160 volt switchgear room (Unit 2)
- Train A dc switchgear room (Unit 2)
- Train B dc switchgear room (Unit 2)

- Train C dc switchgear room (Unit 2)
- Train D dc switchgear room (Unit 2)

The inspectors observed the material condition of plant fire protection equipment, the control of transient combustibles and ignition sources, and the operational status of barriers and reviewed relevant records. The inspectors verified that the rooms contained the fire detection and protection equipment as documented in the updated fire hazards analysis report. The inspectors reviewed Procedure SO123-XV-4.13, "Control of Work and Storage Areas Within the Protected Area," Revision 5. The inspectors reviewed Action Request (AR) 000900505 associated with the transients combustibles in the Train B emergency core cooling system pump room and AR 010300320 associated with transient combustibles in the Train B component cooling water surge tank room.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07B)

a. Inspection Scope

The inspectors reviewed a selected sample of safety-related heat exchanger testing and inspection, cleaning and maintenance records for the component cooling water, shutdown cooling, and EDG heat exchangers. This review was performed to verify that the licensee maintained the heat exchangers in a condition as described in the original plant design in order to perform their safety-related functions. The inspectors also verified that the licensee had identified: 1) potential heat exchanger deficiencies, which could mask degraded performance; and 2) potential common cause heat sink performance problems, which had the potential to increase risk. In addition, the inspectors reviewed heat exchanger design calculations and vendor information for the subject heat exchangers to ensure that the heat exchangers were performing within their design basis.

The inspectors reviewed five ARs initiated by the licensee that addressed problems or deficiencies associated with safety-related heat exchangers to ensure that appropriate corrective actions were being taken.

b. Findings

The inspectors identified an example of a noncited violation (NCV) of very low risk significance pertaining to design control. The inspectors found that the measures the licensee implemented for design control were not adequate in that the design basis of the shutdown cooling heat exchangers was not correctly translated from specifications into procedures, as required by 10 CFR Part 50, Appendix B, Criterion III.

During the review of test results for the shutdown cooling heat exchangers, the inspectors noted that the flow rate on the shell side (component cooling water) was 6980 gpm for the 2ME003 heat exchanger, and 6950 gpm for the 3ME003 heat exchanger. The inspectors observed that Procedure SO23-V-3.26, “Shutdown Cooling Heat Exchanger Testing,” Revision 2, contained a maximum flow limit of 6560 gpm.

This limit was based on a vendor maximum specified in Technical Manual SO23-932-14-0, “Shutdown Cooling Heat Exchanger.” In Section E-3.1, “Design and Operating Conditions,” the vendor states that “[e]quipment must not be operated at pressure or temperature conditions which exceed those specified on the name plate or the flow rates given in the exchanger specification sheet for Mode 1 (design). See Par. E-3.6 for special precautions.”

Section E-3.6, “Special Precautions,” states that “[e]xcessive fluid velocities can result in damages due to erosion and/or vibration. Tube side fluid circulation [primary coolant] should not exceed 5320 gallons per minute for prolonged periods. Shell side fluid circulation [component cooling water] should not exceed 6560 gallons per minute.”

In addition to the technical manual, the licensee had additional information from the vendor that supported the maximum component cooling water flow through the shutdown cooling heat exchangers. The licensee was provided the additional information in Letter S-CE-7634, “Shutdown Cooling System Operational Requirements,” dated June 28, 1982. In Enclosure 1 to this letter, “San Onofre Nuclear Generating Station Units 2 & 3 - Shutdown Cooling System Operation: Requirements for the Upgraded Design,” dated May 1982, Precaution 5.12, states “Do not exceed a SDCHX [shutdown cooling heat exchanger] tube side flow rate of 5320 gpm or a shell side flow rate of 6560 gpm.” (Note that the limitation on tube side flow rate in this document is unconditional, whereas the limitation in the technical manual was applicable only “for prolonged periods.”)

On December 29, 2000, the licensee issued Procedure SO23-3-1.8, “Draining the Reactor Coolant System,” Revision 17. Included in this revision was a change to the flow requirements for the shutdown cooling heat exchangers. In Revision 16 of this procedure, the licensee had stated the flow requirements in step 6.5 of Attachment 21 as they were provided on the exchanger specification sheet for the tube side only (4200 gpm through one heat exchanger). However, Revision 17 replaced those values with the values for both the tube and shell sides (5320 gpm tube side flow through on heat exchanger and 6560 gpm shell side flow through one heat exchanger), as stated in Enclosure 1 to Letter S-CE-7634.

During this inspection period, the licensee was operating the shutdown cooling system on Unit 3 to maintain the unit in Mode 5. The indicated flow on one gauge was pegged at greater than 6500 gpm. Another gauge read approximately 7000 gpm. The inspectors noted that the operating procedures allowed the component cooling water flow through the shell side of the shutdown cooling heat exchangers to be as high as 7500 gpm.

The inspectors found this issue to have a credible impact on safety as a result of the licensee not being able to demonstrate that the shutdown heat exchangers had not been damaged as a result of operating with flows that exceeded the vendor’s specified maximum for a significant amount of time, thereby rendering the heat exchangers unable to perform their design safety functions. The inspectors also found that the loss of the shutdown cooling heat exchangers could credibly affect the operability, availability, reliability, or function of a system or train in a mitigating system. On the basis of these findings, the inspectors assessed the issue using the Significance Determination Process (SDP).

This issue potentially affected the ability to remove decay heat from the reactor. Therefore, the questions for mitigation systems on the Phase 1 screening worksheet were addressed by the inspectors. The inspectors answered each question in the negative because there was not an actual loss of safety function of a system, the Technical Specification allowed outage time was not exceeded for a single train as the result of an actual loss of safety function, there was not an actual loss of safety function for more than 24 hours for equipment designated as risk-significant in accordance with 10 CFR 50.65, and this issue did not involve external events. As a result, this issue was determined to be of very low safety significance (Green).

10 CFR Part 50, Appendix B, Criterion III, states, in part, that "[m]easures shall be established to assure that . . . the design basis . . . are correctly translated into specifications, . . . procedures, and instructions." Contrary to this requirement, the licensee measures for design control were not adequate in that those measures failed to assure that the design basis flow for the shutdown cooling heat exchangers was correctly translated from specifications into procedures.

This example of a violation was of low safety significance because there were no audible indications of excessive flow, and there was not a history of leaks associated with the heat exchangers. This example of a violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. This is one example of a violation (361; 362/2001003-01) and was entered into the licensee's corrective action program as AR 010300419. (The other example is in Section 4OA5.3.)

## 1R11 Licensed Operator Requalification (71111.11)

### a. Inspection Scope

The inspectors observed licensed operator requalification training activities, including the licensed operators' performance and evaluators' critique, and compared performance in the simulator on March 1, 2001, with performance in the control room on March 7. The inspectors observed a simulator scenario based on the Unit 3 loss of offsite power event of February 3, 2001, a replay of that scenario that was frozen several times to facilitate training on rapidly changing indications and conditions, and another scenario involving a steam line break outside of containment in conjunction with an anticipated transient without scram. The inspectors reviewed ARs 010300179, 010300186, and 010300329, all associated with problems in the implementation of the simulator scenarios.

### b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed the implementation of the requirements of the Maintenance Rule (10 CFR 50.65), reviewed AR 010301123, and discussed the Maintenance Rule implications with the cognizant Site Technical Services supervisor for the following functions as the result of the February 3 fire and unit trip in Unit 3:

- Nonsafety-related 4160 volt electrical power (Function MR-4KV-03)
- Main turbine (Function MR-TBN-01)
- Transformers (Function MR-XFRMS-01)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors reviewed the effectiveness of risk assessment and risk management for the following activities:

- EDG 2G002 and Saltwater Cooling (SWC) Pump 2P113 inoperable on March 2, 2001 (Unit 2)

- Auxiliary Feedwater Pump 2P504, SWC Pump 2P113, and Swing Component Cooling Water Pump Breaker 2A0606 inoperable on March 6 (Unit 2)

- Plant Protection System Channel C parameters in bypass, Control Element Assembly Calculator 2 inoperable, and Control Room Emergency Air Cleanup System (CREACUS) B inoperable on March 14 (Unit 2)

- Charging Pump 3P192, SWC Pump 3P114, Auxiliary Feedwater Pump 3P504, and CREACUS B inoperable on March 14 (Unit 3)

- SWC Pumps 2P113 and 2P307, Firewater Pump MP220, and Instrument Air Compressor MC003 inoperable on March 19 (Unit 2)

- SWC Pumps 2P112 and 2P113, Firewater Pump MP220, and the 4 kV crosstie Breakers 2A0417 and 3A0416 inoperable on March 23 (Unit 2)

- EDG 3G002, SWC Pump 3P113, Atmospheric Dump Valve 3HV8419, Firewater Pump MP222, inoperable on March 26 (Unit 3)

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the operability evaluations documented in the following ARs to ensure the operability was properly justified:

- Abnormal start of Turbine Driven Auxiliary Feedwater Pump 2P140 (AR 001201078) (Unit 2)

- Pressurizer Spray Check Valve S21201MU976 inservice test failure (AR 010100281) (Unit 2). In addition to reviewing the documented operability assessment, the inspectors discussed the licensee's ongoing investigation into testing problems involving a similar Unit 3 valve (S31201MU977) with the manager leading the licensee's investigation.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed and/or reviewed postmaintenance testing for the following activities to verify that the test procedures and activities adequately demonstrated system operability:

- Auxiliary feedwater discharge to Steam Generator 3E088 Isolation Valve 3HV4714 repairs. (AR 010201279 and Maintenance Order (MO) 01021961000) (Unit 3).

- Replacement of low temperature overpressure Relief Valve 3PSV9349 (MOs 99083312000 and 99083312000; Procedure SO23-3-2.6, "Shutdown Cooling System Operation," Temporary Change Notice 17-1, Attachment 1; Procedure SO23-3-2.7.2, "Safety Injection System Removal/Return to Service Operation," Temporary Change Notice 0-1, Attachment 6; Procedure SO23-3-2.7.2, Temporary Change Notice 6-2, Attachments 4 and 5; and Procedure SO23-I-2.59, "Shutdown Cooling System Relief Valve Surveillance," Revision 7) (Unit 3)

- Replace Inverter 3EY002 supply breaker (MOs 00040753000 and 99050197000 and Procedure SO23-6-17, "Class 1E 120 VAC Vital Bus Power Supply System Operation," Revision 10, Attachments 3 and 4) (Unit 3)

- Clean, inspect, and adjust auxiliary relay panel for auxiliary feedwater Valve 3HV4705 (Work Authorization Record 3-R1AFW02; MO 99082443000; Procedure SO23-3-3.30.6, "Auxiliary Feedwater System Online Valve Test," Temporary Change Notice 5-1, Attachment 1; Operating Instruction SO23-2-4, "Auxiliary Feedwater System Operation," Revision 18, Attachment 3; Procedure SO23-3-3.43.30, "ESF Subgroup Relays K-112A, K-625A and K-725A Semiannual Test," Revision 3; and Procedure SO23-3-3.43.42, "ESF Subgroup Relays K-523A and K-723A Semiannual Test," Revision 3) (Unit 3)

- Charging Pump 2P192 repack, lubrication, and inspection (MOs 00111022000 and 01011583000, Work Authorization Record 2-0100244, Procedure SO23-3-3.60.5, "Charging Pump and Valve Testing," Revision 3, Attachment 4; and Pump 2P192 inservice pump test record) (Unit 2)

- Auxiliary Feedwater Pump 2P504 breaker replacement (MO 01030072000, AR 010300347, and control room logs dated March 3, 2001) (Unit 2)

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

During the Unit 3 forced outage, the inspectors periodically monitored operational status of the shutdown cooling system and the vital and nonvital electrical power distribution systems. The inspectors confirmed that the licensee's monitoring for the potential buildup of noncondensible gases in the reactor vessel head was adequate. The inspectors toured the containment to ensure that activities were being conducted as planned.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed and/or reviewed documentation for the following surveillance tests to verify that the structures, systems, and components are capable of performing their intended safety functions and to assess their operational readiness:

- EDG 3G002 automatic start on a loss of offsite power. The inspectors observed the performance of the surveillance test, reviewed Procedure SO23-3-3.23.1, "Diesel Generator Refueling Interval Tests," Revision 16, and discussed the performance of the test with Operation's supervision.

- The inspectors reviewed testing that the licensee performed to satisfy the requirements of Technical Specification Surveillance Requirement 3.8.1, specifically, how the licensee addressed the testing of the EDG automatic voltage regulators. The inspectors reviewed the licensee's assessment of the requirements as documented in AR 991000186.

- The inspectors reviewed the surveillance testing requirements for the shunt trip portion of the reactor trip circuit breakers. The inspectors reviewed AR 010201704, control room operator logs for February 28, 2001, Drawing 8052-B4.4, and discussed the testing with a Station Technical Instrumentation and Control supervisor.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Temporary Facility Modification 2-00-BBA-001 associated with deleting the intergasket alarm for reactor coolant Pump 2P002, and Procedure SO123-V-5.10, "Temporary Facility Modification (TFM)," Revision 8. In addition, the inspectors reviewed AR 001101113 and its associated 10 CFR 50.59 evaluation.

b. Findings

No findings of significance were identified.

**2. RADIATION SAFETY**
Cornerstone: Occupational Radiation Safety

2OS2 ALARA (as low as reasonably achievable) Planning and Controls (71121.02)

a. Inspection Scope

The inspectors interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs in the radiologically controlled areas. Independent radiation surveys of selected work areas within the radiologically controlled area were conducted. The following items were reviewed and compared with regulatory requirements to determine whether the licensee had an adequate program to maintain occupational exposure ALARA:

- ALARA program procedures
- Nuclear Oversight Division ALARA Program Surveillance SOS-018-00
- Health Physics Division Directed Assessment of U2C11 ALARA Planning and Controls

- Health Physics Division Self-Assessment of U2C11 and U3C11 Refueling Outages
- Health Physics Division Fourth Quarter 2000 Self-Assessment SO123-SA-1
- Processes used to estimate and track exposures
- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average dose information
- Three ALARA work activity packages for the Unit 3 Cycle 11 Refueling Outage (steam generator primary side work, reactor disassembly/reassembly, and pressurizer heater replacement) which resulted in the highest personnel collective exposures during the inspection period
- Use of engineering controls to achieve dose reductions
- Individual exposures of selected work groups (Health Physics, Operations, and Maintenance)
- Plant-related source term data, including source term control strategy
- Hot spot tracking and reduction program
- Temporary shielding packages (TSA 01-002, 01-007, 01-014, 01-016, and 01-028)
- Radiological work planning and ALARA prejob briefings for two work activities scheduled during the inspection week (Unit 2 containment entry and removal of Unit 1 fuel handling tool from the Unit 2 spent fuel pool)
- Job site inspection and ALARA controls for one work activity (reactor coolant pump rebuild)
- ALARA Committee meeting minutes (5/19/00, 9/7/00, 12/8/00, and 3/22/01)
- Declared pregnant worker dose monitoring controls
- A summary of ALARA-related ARs written since the last inspection in this area. Twelve of these ARs, which involved ALARA concerns during the U3C11 refueling outage, were reviewed in detail (001201166, 001200930, 010100304, 010100618, 010100623, 010101112, 010101386, 010101984, 010102096, 010102461, 010200991, and 010201061).

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation (71121.03)

a. Inspection Scope

The inspectors interviewed cognizant licensee personnel and reviewed the following items to ensure that the licensee's activities met regulatory requirements.

- Calibration, operability, and alarm setpoints, when applicable, of portable radiation detection instrumentation, whole-body counting instrumentation, temporary area radiation monitors, continuous air monitors, electronic alarming dosimeters, and personnel contamination monitors

- Calibration, operability, and alarm setpoints, when applicable, of area radiation monitors not covered by the maintenance rule

- Calibration expiration and source response performance test currency on radiation detection instruments staged for use

- Health Physics technician instrument selection and self-verification of instrument operability prior to use

- The status and surveillance records of self-contained breathing apparatuses staged and ready for use in the plant

- The licensee's capability for refilling and transporting self-contained breathing apparatus air bottles to and from staged plant locations (i.e., the control room and Operations Support Center) and the bottled air refilling facility located at the Mesa location during emergency conditions

- Control room operator and Health Physics emergency response personnel training and qualifications for use of self-contained breathing apparatus

- Selected exposure significant radiological incidents that involved radiation monitoring instruments or self-contained breathing apparatus deficiencies

- One audit and six self-assessments (Quality Assurance Audit SCES-909-99 and Health Physics Division quarterly self-assessments for the third quarter 1999 through fourth quarter 2000)

- Health Physics procedures implementing the radiation instrumentation program and respiratory protection program

- A summary of radiological ARs written between July 1, 1999, and February 9, 2001. The following twelve ARs were reviewed in detail: 990701585, 990900770, 991100493, 991100571, 991100778, 991101176, 000201133, 000201180, 000901194, 010100297, 010101102, and 010200664.

b. Findings

No findings of significance were identified.

## 4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors verified the accuracy of data reported by the licensee for the following performance indicators to ensure that the performance indicator color was correct:

- IE3 Unplanned Power Changes (Unit 2)
- IE3 Unplanned Power Changes (Unit 3)

The inspectors reviewed data from the plant monitoring system for all of calendar year 2000 and focused particular attention on a Unit 2 power reduction on December 23, 2000. The data points reviewed for this transient included:

| | |
|---|---|
| CV5993 | Smoothed Plant Power |
| CV9000 | Plant Power |
| CV9005 | Secondary Calorimetric Power |
| CV9005AV | Average Secondary Calorimetric Power |
| CV9006 | Turbine Power |
| CV9615 | Secondary Calorimetric Power Based o Feedwater Flow |
| CV9616 | Secondary Calorimetric Power Based on Steam Flow |
| CV9739 | Core Delta T Power |
| J001A | Neutron Power Channel A |
| J001B | Neutron Power Channel B |

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

(Closed) Licensee Event Report 361; 362/2000-010-00: CREACUS not seismically qualified.

The licensee identified a cracked weld on the electrical power supply conduit coupling connection to the sheet metal junction box for the CREACUS Train B recirculation fan. The licensee was unable to determine the cause of the cracking.

The licensee determined that the power supply could have shorted out during a seismic event, preventing the CREACUS Train B from performing its safety function (to protect the control room environment from a toxic or radioactive gas hazard). The issue was characterized as a Green finding (very low safety significance) using the mitigating systems cornerstone SDP (FIN 361; 362/2001003-02), because only one cornerstone was involved, only one train was affected, and there was no actual loss of safety function. This issue is in the licensee's corrective action program as AR 000801751.

The licensee determined that the condition was a violation of Technical Specification 3.7.11, because the condition probably existed for longer than the 7-day allowed outage time, and the licensee had not implemented the required actions. However, the licensee corrected the problem within 3 days from the time the condition was identified, in compliance with the Technical Specification required actions; therefore, no violation of NRC requirements occurred.

4OA5 Other

.1 Third-Party Evaluations

The inspectors reviewed the report of a peer evaluation of San Onofre that had been performed by the World Association of Nuclear Operators. The 2-week evaluation had been conducted in June 2000 and the report was issued in March 2001.

.2 Financial Status

The NRC has exercised communications channels to better understand the licensee's planned and implemented actions, especially as they relate to safely operating the reactors. The inspectors have specifically reviewed the following on a weekly basis:

- Staffing of on-shift operating personnel and the number of qualified Emergency Response Organization responders
- The corrective maintenance backlog
- The corrective actions backlog
- Reduction in safety or risk important outage activities
- Reduction in planned risk important modifications or enhancements
- Emergency Response Facility and siren availability
- Generator voltage loading
- Impact of rolling blackouts of the grid on offsite power availability

NRC inspections and inspector observations, to date, have confirmed that the licensee operated the units safely and that public health and safety was, thus far, assured.

.3 <u>(Closed) Unresolved Item 361; 362/2000003-03</u>: verification of chilled water flow through safety-related room coolers.

During the performance of the Safety System Design and Performance Capability inspection, an NRC team questioned the capability of safety-related room coolers, in each unit, to perform their intended safety function. At the time of the inspection, the team did not have enough information to determine if the safety-related room coolers had adequate flow from the emergency chilled water system in order to perform their intended safety functions.

As noted in NRC Inspection Report 50-361; 362/00-03, the licensee had elected to perform testing to demonstrate the capability of the safety-related room coolers. However, that testing was identified by the NRC team to be inadequate for this purpose.

Subsequent to the inspection, the licensee initiated a plan to gather data to develop a correlation between flow through the coolers and the differential pressure across the coolers. The inspectors noted that the preliminary results from two tests indicated that the vendor supplied data underestimated the flow at a given differential pressure. For example, at a differential pressure of 8 inches water column, the vendor stated that the flow should be approximately 4.75 gpm. The licensee measured approximately 5.7 gpm. On the basis of the data collected by the licensee, the inspectors could not determine if the flow to each room cooler was adequate. The data only indicated that the flow may be more than required to remove the required heat, but there was no indication that the flow did not exceed the maximum allowable flow the coolers were designed to carry.

While the data gathered to date were not conclusive, the data have demonstrated that the licensee did not have the information necessary to demonstrate that the safety-related room coolers were capable of performing their intended functions. Consequently, the measures developed by the licensee to meet the requirements of Criterion III were not adequate in that the data to demonstrate adequate flow balance for the emergency chilled water system were not available.

The inspectors found this issue to have a credible impact on safety as a result of the licensee not being able to demonstrate the capability of safety-related room coolers that, if they should not have adequate chilled water flow, could reasonably result in the loss of safety-related equipment used for mitigating an accident. The inspectors also found that the loss of the safety-related room coolers could credibly affect the operability, availability, reliability, or function of a system or train in a mitigating system. On the basis of these findings, the inspectors assessed the issue using the SDP.

This issue potentially affected all mitigating systems since the safety-related room coolers provided cooling for the mitigating equipment. Therefore, the questions for mitigation systems on the Phase 1 screening worksheet were addressed by the inspectors. The inspectors answered each question in the negative because there was not an actual loss of safety function of a system, the Technical Specification allowed outage time was not exceeded for a single train as the result of an actual loss of safety function, there was not an actual loss of safety function for more than 24 hours for equipment designated as risk-significant in accordance with 10 CFR 50.65, and this issue did not involve external events. As a result, this issue was determined to be of very low safety significance.

10 CFR Part 50, Appendix B, Criterion III, states, in part, that "[m]easures shall be established to assure that . . . the design basis . . . are correctly translated into specifications, . . . procedures, and instructions." Further, Criterion III states, in part, that "design control measures shall provide for verifying or checking the adequacy of design, such as . . . by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program."

The failure to have adequate measures to control the design basis was identified as another example of a violation of 10 CFR Part 50, Appendix B, Criterion III (361; 362/2001003-01). This example of the violation was of low safety significance (Green) because the total emergency chilled water flow exceeded the system design basis, and the preliminary test data, along with calculations, provide assurance that adequate flow can be supplied to each safety-related room cooler. This example of the violation was entered into the licensee's corrective action program as AR 000401144. (The other example is in Section 1R07.)

.4 (Closed) Inspection Followup Item 361; 362/1998009-01: review of corrective actions taken for containment high range radiation monitor accuracy.

AR 970301240, initiated March 27, 1997, described problems regarding accuracy of the containment high range radiation monitors. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 2, states, in part, that the containment high range monitor detectors should respond to gamma radiation with an overall dose rate system accuracy within a factor of 2 over the entire instrument range.

In June 1996 the licensee determined that total loop uncertainty calculations for the containment high range monitor during loss of coolant accident tests showed that time dependent factors caused errors in system accuracy beyond a factor of 2 as specified in Regulatory Guide 1.97. To correct this condition, the licensee replaced the high range radiation monitors' coaxial cable with mineral-insulated cable. The licensee also revised the total loop uncertainty calculations for the containment high range radiation monitors to more accurately predict the behavior of the monitors during postaccident conditions. Based on the results of the revised total loop uncertainty calculations, the Updated Final Safety Analysis Report was revised to incorporate the methodology and results of the revised calculation used to comply with Regulatory Guide 1.97. The inspectors reviewed the completion of the licensee's corrective actions and determined them to be satisfactory. This inspection followup item is closed.

4OA6 Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. R. Krieger and other members of licensee management at exit meetings on March 2, 8, and 23 and on April 3, 2001. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. While information marked as proprietary was reviewed by the inspectors, no proprietary information was included in the report.

# ATTACHMENT

## SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Allen, Supervisor, Reliability Engineering
C. Anderson, Manager, Site Emergency Preparedness
D. Axline, Engineer, Licensing
J. Barrow, ALARA Project Manager, Health Physics
D. Brieg, Manager, Station Technical
G. Cook, Supervisor, Nuclear Oversight and Regulatory Affairs
B. Corbett, Supervisor, Health Physics
M. Farmer, General Foreman, Health Physics
J. Fee, Manager, Maintenance
M. Grove, General Foreman, Health Physics
J. Hirsch, Manager, Chemistry
M. Humphrey, Supervisor, Health Physics Instrumentation
R. Krieger, Vice President, Nuclear Generation
J. Madigan, Manager, Health Physics
M. McBrearty, Engineer, Nuclear Oversight and Regulatory Affairs
D. Nunn, Vice President, Engineering and Technical Services
N. Quigley, Manager, Station Technical Services
R. Richter, Supervisor, Fire Protection Engineering
A. Scherer, Manager, Nuclear Oversight and Regulatory Affairs
S. Schofield, Supervisor, Health Physics Self Assessment
M. Short, Manager, Site Technical Support
T. Vogt, Plant Superintendent, Units 2 and 3 Operations
R. Waldo, Manager, Operations
R. Warnock, Supervisor, Health Physics Instrumentation/Dosimetry

NRC

Gail Good, Chief, Plant Support Branch, Region IV
Claude Johnson, Chief, Engineering and Maintenance Branch, Region IV

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed During this Inspection

| | | |
|---|---|---|
| 361; 362/2001003-01 | NCV | inadequate measures to assure that design basis information is correctly translated and maintained (Sections 1R07 and 4OA5.3) |
| 361; 362/2001003-02 | FIN | CREACUS not seismically qualified (Section 4OA3) |

Previous Items Closed

| | | |
|---|---|---|
| 361; 362/2000-010-00 | LER | CREACUS not seismically qualified (Section 4OA3) |
| 361; 362/2000003-03 | URI | verification of chilled water flow through safety-related room coolers (Section 4OA5.3) |
| 361; 362/1998009-01 | IFI | review of corrective actions taken for containment high range monitor accuracy (Section 4OA5.4) |

## LIST OF ACRONYMS USED

| | |
|---|---|
| ALARA | as low as reasonably achievable |
| AR | action request |
| CFR | Code of Federal Regulations |
| CREACUS | control room emergency air cleanup system |
| EDG | emergency diesel generator |
| IFI | inspection followup item |
| LER | licensee event report |
| NCV | noncited violation |
| NRC | Nuclear Regulatory Commission |
| MO | maintenance order |
| SDP | Significance Determination Process |
| SWC | saltwater cooling |
| URI | unresolved item |

## PARTIAL LIST OF DOCUMENTS REVIEWED

Action Requests

| | | |
|---|---|---|
| 990900499 | 000300959 | 010102118 |
| 991000590 | 000401144 | 010200072 |

Drawings

| NUMBER | TITLE | REVISION |
|---|---|---|
| 40110CSO3 | P&I Diagram Diesel Generator System (Train B) System 2420 | 23 |
| 40110D | P&I Diagram Diesel Generator System (Train B) System 2420 | 26 |
| 40126A | P&I Diagram Component Cooling Water System (Salt Water Pumps) System 1203 | 21 |

| NUMBER | TITLE | REVISION |
|---|---|---|
| 40126B | P&I Diagram Component Cooling Water System (Salt Water Pumps) System 1203 | 20 |
| 40127A | P&I Diagram Component Cooling Water System (Pumps) System 1203 | 21 |
| 40127C | P&I Diagram Component Cooling Water System (Heat Exchangers) System 1203 | 35 |
| DBD-SO23-750, Figure B-1 | Emergency Diesel Generators, Rating at Elevated Temperature (°F) for EMD 645E1, 38, E4B, E9B Engines | 1 |
| SO23-403-12-59 | Cooling Water Schematic | 13 |

Calculations

| NUMBER | TITLE | REVISION |
|---|---|---|
| M-0076-040 | Diesel Generator Building Emergency Cooling Load Calculation | 0; 2, CCN 1 |
| M-0076-041 | Diesel Generator Building - Emergency Equipment Sizing | 0, CCN 2 |
| M-76-42 | Diesel Generator Building - Emergency Duct Sizing Calculations | 0 |
| M-76-43 | Thermal Analysis of Diesel Generator Radiator | 1 |

Miscellaneous Documents

| NUMBER | TITLE | REVISION |
|---|---|---|
| SD-SO23-400 | Component Cooling Water System | 6 |
| SD-SO23-740 | Safety Injection, Containment Spray, and Shutdown Cooling System | 8 |
| SD-SO23-750 | Emergency Diesel Generators | 7 |

Procedures

| NUMBER | TITLE | REVISION |
|---|---|---|
| SO23-V-2.11 | Generic Letter 89-13 Commitments | 0b |
| SO23-V-3.25 | Component Cooling Water Heat Exchanger Testing | 6, EC 6-1 |
| SO23-V-3.26 | Shutdown Cooling Heat Exchanger Testing | 2 |